UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 30, 2007

EA-07-095

Southern Nuclear Operating Company, Inc.
ATTN: Mr. T. E. Tynan
Vice President - Vogtle
Vogtle Electric Generating Plant
7821 River Road
Waynesboro, GA 30830

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT- NRC INTEGRATED INSPECTION REPORT 05000424/2007002 AND 05000425/2007002, ANNUAL ASSESSMENT MEETING SUMMARY, AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Tynan:

On March 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Vogtle Electric Generating Plant, Units 1 and 2. The enclosed integrated inspection report documents the inspection findings, which were discussed on April 12, 2007, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one self-revealing finding of very low safety significance (Green) was identified. In addition, one licensee-identified violation, which was determined to be of very low safety significance, is listed in the enclosed inspection report. The NRC is treating this violation as a non-cited (NCV) violation consistent with Section VI.A of the NRC Enforcement Policy. If you contest this non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Vogtle.

Also, it was determined that materially inaccurate and incomplete information was identified in the mitigating systems performance index (MSPI) data submitted to the NRC. NRC inspectors identified an additional 20.5 hours of unavailability for the Unit 2 Nuclear Service Cooling Water system which had not been included in the Unit 2 Cooling Water Systems MSPI. Your analysis indicated that the additional hours would not change the color of any previously reported data; however, the licensee did not review the effect the additional hours would have on future MSPI reporting. On January 17, 2007, you submitted the fourth quarter 2006 Unit 2 Cooling Water

Systems MSPI as being Green. On February 1, you determined that the additional hours caused the MSPI to change to White. On February 7, you submitted a correction to the NRC and reported the MSPI as White. As such, the NRC has determined that a Severity Level IV violation of 10 CFR 50.9, Completeness and Accuracy of Information, occurred. This violation was evaluated in accordance with the Enforcement Policy, which is included on the NRC's Web site at *www.nrc.gov*. However, after consultation with the Director, Office of Enforcement, I have been authorized to exercise enforcement discretion pursuant to Section VII.B.6, Violations Involving Special Circumstances, of the Enforcement Policy to refrain from issuing a Notice of Violation. Discretion is warranted in this case (1) because submission of the incomplete and inaccurate MSPI information was not willful; (2) because the incomplete and inaccurate MSPI information was identified within a period of 1 year after the beginning of MSPI data collection, or by April 1, 2007; and (3) in recognition of (a) ongoing PI development activities, (b) the time constraints to gather and submit historical data, (c) the large volume of data (12 quarters of data) needed to calculate and verify the MSPIs, and (d) the time needed for licensees to familiarize and adjust to the new reporting guidance.

In accordance with the Code of Federal Regulations 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Charles A. Casto, Director
Division of Reactor Projects

Docket Nos. 50-424, 50-425
License Nos. NPF-68 and NPF-81

Enclosure: Inspection Report 05000424/2007002 and
05000425/2007002
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Charles A. Casto, Director
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Letter to T. E. Tynan from Scott M Shaeffer dated April 30, 2007
SUBJECT: VOGTLE ELECTRIC GENERATING PLANT- NRC INTEGRATED INSPECTION REPORT 05000424/2007002 AND 05000425/2007002, ANNUAL ASSESSMENT MEETING SUMMARY, AND EXERCISE OF ENFORCEMENT DISCRETION

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-424, 50-425
License Nos.:	NPF-68, NPF-81
Report Nos.:	05000424/2007002 and 05000425/2007002
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Vogtle Electric Generating Plant, Units 1 and 2
Location:	Waynesboro, GA 30830
Dates:	January 1, 2007 through March 31, 2007
Inspectors:	G. McCoy, Senior Resident Inspector B. Anderson, Resident Inspector C. Even, Reactor Inspector (Sections 1R02 and 1R17) R. Lewis, Reactor Inspector (Sections 1R02 and 1R17) E. Michel, Reactor Inspector (Sections 1R02 and 1R17) B. Miller, Reactor Inspector (Section 4OA5) R. Rodriguez, Reactor Inspector (Sections 1R02 and 1R17) M. Scott, Senior Reactor Inspector (Sections 1R02 and 1R17) A. Vargas - Mendez, Reactor Inspector (Section 1R08)
Accompanying Personnel:	S. Parra
Approved by:	Scott Shaeffer, Chief Reactor Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000424/2007-002, 05000425/2007-002; 01/01/2007 - 03/31/2007; Vogtle Electric Generating Plant, Units 1 and 2; Event Followup.

The report covered a three-month period of inspection by two resident inspectors and seven reactor inspectors. One Green finding was identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding was identified for poor work practices associated with the installation of the loop 2 main feedwater regulating valve (MFRV) electro-pneumatic transducer. The transducer was mounted in a manner that did not prevent water ingress and allow drainage. During washdown activities in the vicinity of the MFRV water entered the transducer causing the loop 2 MFRV to close. When the MFRV could not re-opened from the control room, operators manually tripped the reactor.

  This finding is more than minor because it affected the procedure quality attribute of the Initiating Events cornerstone in that inadequate installation instructions caused transducer failure which resulted in a manual reactor trip. The finding was determined to be of very low safety significance (Green) because it did not increase the likelihood that mitigation equipment or functions would not be available. This finding is directly related to the complete and accurate procedures aspect of the Human Performance cross-cutting area because the procedure did not incorporate the vendor's recommendations. (Section 4OA3.2)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation is listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at full rated thermal power (RTP) for the entire inspection period.

Unit 2 started the inspection period at full RTP. The unit was reduced to 75% RTP on February 20 to perform repairs on nuclear power instrument 2NI-43. Following these repairs, Unit 2 was returned to full RTP. The unit was shutdown on March 4 for a planned refueling outage.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluations of Changes, Tests or Experiments

a. Inspection Scope

The inspectors reviewed selected samples of evaluations to confirm that the licensee had appropriately considered the conditions under which changes to the facility, Updated Final Safety Analysis Report (UFSAR), or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed evaluations for seven changes and additional information, such as calculations, supporting analyses, the UFSAR, and drawings to confirm that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The seven evaluations reviewed are listed in the Attachment.

The inspectors also reviewed samples of changes for which the licensee had determined that evaluations were not required, to confirm that the licensee's conclusions to "screen out" these changes were correct and consistent with 10CFR50.59. The sixteen "screened out" changes reviewed are listed in the Attachment.

The inspectors reviewed Condition Reports (CRs) to verify that problems were identified at an appropriate threshold, were entered into the corrective action program, and appropriate corrective actions had been initiated.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial Walkdowns. The inspectors performed partial walkdowns of the following two systems to verify correct system alignment. The inspectors checked for correct valve and electrical power alignments by comparing positions of valves, switches, and breakers to the procedures and drawings listed in the Attachment. Additionally, the inspectors reviewed the CR database to verify that equipment alignment problems were being identified and appropriately resolved.

- Unit 1 auxiliary feedwater (AFW) system during Unit 1 train B diesel generator extended maintenance outage
- Unit 1 train B diesel generator during Unit 1 train A diesel generator extended maintenance outage

Complete System Walkdown. The inspectors performed a complete walkdown of the Unit 1 charging system. The inspectors performed a detailed check of valve positions, electrical breaker positions, and operating switch positions to evaluate the operability of the redundant trains or components by comparing the required position in the system operating procedure to the actual position. The inspectors also reviewed control room logs and CRs to verify that alignment and equipment discrepancies were being identified and appropriately resolved. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

Fire Drill Observation. On February 5, inspectors observed a fire drill conducted in the control building, level A, room RA-59. The inspectors assessed the adequacy of the fire drill and fire brigade response using licensee procedures 92000-C, Fire Protection Program; 92005-C, Fire Response Procedure; 92030-C, Fire Drill Program; 92786-1, Zone 86 - Control Building - Level A Fire Fighting Preplan; and 17103A-C, Annunciator Response Procedures for the Fire Alarm Computer. The inspectors evaluated the fire brigade performance to verify that they responded to the fire in a timely manner, donned proper protective clothing, used self-contained breathing apparatus, and had the equipment necessary to control and extinguish the fire. The inspectors assessed the adequacy of the fire brigade's fire fighting strategy including entry into the fire area, communications, search and rescue, and equipment usage.

Fire Area Tours. The inspectors walked down the following eight plant areas to verify the licensee was controlling combustible materials and ignition sources as required by procedures 92015-C, Use, Control, and Storage of Flammable/Combustible Materials, and 92020-C, Control of Ignition Sources. The inspectors assessed the observable condition of fire detection, suppression, and protection systems and reviewed the licensee's fire protection Limiting Condition for Operation log and CR database to verify that the corrective actions for degraded equipment were identified and appropriately prioritized. The inspectors also reviewed the licensee's fire protection program to verify the requirements of UFSAR Section 9.5.1, Fire Protection Program, and Appendix 9A, Fire Hazards Analysis, were met. Documents reviewed are listed in the Attachment.

- Unit 1 train B diesel generator (DG) building and associated fuel oil day tank room
- Unit 2 train A and train B 4160 Volt switchgear rooms
- Unit 1 train A and train B nuclear service cooling water (NSCW) pumphouses
- Unit 2 AFW Building
- Unit 1 Engineered Safety Features (ESF) chiller rooms

- Unit 2 ESF chiller rooms
- Unit 2 train B cable spreading room
- Unit 2 train A and train B NSCW pumphouses

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

a. Inspection Scope

The inspectors observed in-process ISI work activities, reviewed ISI procedures, and reviewed selected ISI records associated with risk significant structures, systems, and components during the Unit 2 refueling outage. The observations and records were compared to the requirements specified in the Technical Specifications (TSs) and the ASME Boiler and Pressure Vessel Code, 1998 Edition, to verify compliance and to ensure that examination results were appropriately evaluated and dispositioned.

The inspectors conducted an onsite review of nondestructive examination (NDE) activities to evaluate compliance with TSs, ASME Section XI, and ASME Section V (1989 Edition) requirements to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of ASME Section XI, IWB-3000 or IWC-3000 acceptance standards. The inspectors observed the following examinations and reviewed the following non-destructive examination (NDE) records.

Ultrasonic Testing (UT)
- Reactor Vessel Head Studs, Nos. 5, 6, and 8
- Feed Water Pipe to Valve, Weld No. 21305-062-6

Penetrant Testing (PT)
- Residual Heat Removal (RHR) 14" Outlet Nozzle to Tube Side Shell Weld Nos. 21205-E6-002-W04 and 21205-E6-002-W05

Magnetic Particle Testing (MT)
- Reactor Vessel Head Support Lugs, Nos. 21201-V6-001W204 and 21201-V6-001W206

Automated Ultrasonic Testing (AUT)
- Reactor Vessel Lower Meridional Weld at 270°, Weld No. W24
- Reactor Vessel Intermediate to Lower Circumferential Weld No. W05
- Safe End Inlet Nozzle Dissimilar Metal (DM) Weld at 67° Inlet Elbow Weld at 67°, No. W34-010-7
- Safe End Outlet Nozzle DM Weld at 158° Outlet Pipe Weld at 158°, No. W39-011-8
- Inlet Nozzle to Shell Weld at 270°, No. W30
- Outlet Nozzle to Shell Weld at 202°, No. W29

The inspectors reviewed the following examination records that contained recordable indications.

- UT: No. 2-CV-110, Chemical Volume Control System (CVCS) Flow Orifice
- VT: No. 21201-B6-001-I10 and 21201-B6-003-I10, Channel Head Drain Tube for Steam Generators (SGs) 1 and 3
- Reactor Vessel Intermediate Long Seam at 240°, Weld No. W17
- Reactor Vessel Lower Long Seam, Weld No.W18
- Reactor Vessel Long Seam, Weld No. W20

Qualification and certification records for examiners, inspection equipment, and consumables along with the applicable NDE procedures for the previously referenced ISI examination activities were reviewed and compared to requirements stated in ASME Section V, ASME Section XI, and other industry standards.

The inspectors reviewed welding activities from the previous outage. The inspectors reviewed drawings, work instructions, weld process sheets, weld travelers, pre-heat requirements and radiography records for welding of an ASME Class 2 pressure boundary weld. The inspectors also reviewed the following NDE records and observed weld overlay activities associated with the Pressurizer Safety Nozzles, Pressurizer Spray Nozzle, Pressurizer Surge Nozzle and the Pressurizer Pressure-Operated Relief Valve (PORV) Nozzle.

<u>PT</u>
- Spray Nozzle N–5, Weld Nos. 21201-V6-W21 and 21201-030-49
- Surge Nozzle N–6, Weld Nos. 21201-V6-002W22 and 21201-053-6
- Safety Nozzle N–1, Weld Nos. 21201-V6-002-W18 and 21201-056-1
- PORV Nozzle N–4, Weld Nos. 21201-V6-0032-W17 and 21201-059-1

The inspectors reviewed the SG examination scope, expansion criteria, eddy current testing (ET) acquisition procedures, ET analysis procedures, the SG Operational Assessment, in-situ tube pressure testing procedures, and records and examination reports to confirm that:

- The SG tube ET examination scope was sufficient to identify tube degradation confirming that the ET scope completed was consistent with the licensee’s procedures and plant TS requirements. In addition, the inspectors reviewed the SG tube ET examination scope to determine that it was consistent with that recommended in EPRI “Pressurized Water Reactor Steam Generator Examination Guidelines,” Revision 6, and included tube areas which represent ET challenges, such as the tubesheet regions, expansion transitions and support plates.

- The ET probes and equipment configurations used to acquire ET data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, “Performance Demonstration for Eddy Current Examination,” of EPRI “Pressurized Water Reactor Steam Generator Examination Guidelines,” Revision 6.

- The licensee adequately evaluated for any contractor deviations from their ET data acquisition or analysis procedures or EPRI "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6. Documents reviewed are listed in the Attachment.

The inspectors performed a review of SG ISI-related problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. In addition, the inspectors' review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspectors evaluated the threshold for identifying issues through interviews with licensee staff and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements.

Boric Acid Corrosion Control (BACC). The inspectors reviewed the licensee's BACC program to ensure compliance with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The inspectors conducted an on-site record review as well as an independent walkdown of parts of the reactor building that are not normally accessible during at-power operations to evaluate compliance with licensee BACC program requirements and 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The inspectors reviewed records to verify that visual examinations focused on locations where boric acid leaks can cause degradation of safety-significant components and that degraded or non-conforming conditions were properly identified in the licensee's corrective action system. The inspectors reviewed a sample of engineering evaluations completed for boric acid found on reactor coolant system piping and components to verify that the minimum design code-required section thickness had been maintained for the affected components. The inspectors also reviewed CRs and corrosion assessments to confirm that they were consistent with BACC program requirements. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors evaluated operator performance on January 25, during licensed operator simulator training described on simulator exercise guide V-RQ-SE-07101. The simulator scenario covered operator actions resulting from a steam generator tube rupture. Procedures reviewed are listed in the attachment. The inspectors specifically assessed the following areas:

- Correct use of the abnormal and emergency operating procedures

- Ability to identify and implement appropriate actions in accordance with the requirements of the Technical Specifications
- Clarity and formality of communications in accordance with procedure 10000-C, Conduct of Operations
- Proper control board manipulations including critical operator actions
- Quality of supervisory command and control
- Effectiveness of post-evaluation critique

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the following two equipment problems to evaluate the effectiveness of the licensee's handling of equipment performance problems and to verify the licensee's maintenance efforts met the requirements of 10 CFR 50.65 (the Maintenance Rule) and licensee procedure 50028-C, Engineering Maintenance Rule Implementation. The reviews included adequacy of the licensee's failure characterization, establishment of performance criteria or 50.65(a)(1) performance goals, and adequacy of corrective actions. Other documents reviewed during this inspection included control room logs, system health reports, the maintenance rule database, and maintenance work orders (MWOs). Also, the inspectors interviewed system engineers and the maintenance rule coordinator to assess the accuracy of identified performance deficiencies and extent of condition.

- CR 2006112573, Slow response of Unit 1 train B DG when the stop button was depressed
- CR 2006111712, Failure of the Unit 1 containment spray containment suction valve 2HV9002A to open from the control room

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the following five risk significant and emergent MWOs to verify plant risk was properly assessed by the licensee prior to conducting the activities. The inspectors reviewed risk assessments and risk management controls implemented for these activities to verify they were completed in accordance with procedure 00354-C, Maintenance Scheduling, and 10 CFR 50.65(a)(4). The inspectors also reviewed the CR database to verify that maintenance risk assessment problems were being identified at the appropriate level, entered into the corrective action program, and appropriately resolved.

- Unit 1 train B DG extended maintenance outage
- Unit 2 train A centrifugal charging pump (CCP) maintenance outage
- Unit 2 NSCW train B tower fan motor #4 replacement
- Unit 1 and Unit 2 offsite power source relay testing
- Unit 2 train A RHR system outage

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following four evaluations to verify they met the requirements of procedure NMP-GM-002, Corrective Action Program, and NMP-GM-002-001, Corrective Action Program Instructions. This included a review of the technical adequacy of the evaluations, the adequacy of compensatory measures, and the impact on continued plant operation.

- CR 2007100824, Increased outboard seal leakage on 2A CCP
- CR 2007100843, Sample of 2PV3000 valve actuator fluid failed cleanliness requirements
- CR 2007101315, Failed ‘C’ processor on 2B sequencer
- CR 2007100077, Operability of the Combustion Turbine in support of 1B DG extended outage

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed engineering change packages for the following eight modifications to evaluate the modifications for adverse effects on system availability, reliability, and functional capability. Documents reviewed are listed in the Attachment.

- MDC 1019003501, AFW Controls Upgrade, Rev. 0 (Mitigating Systems)
  - Testing (overspeed and IST)
  - Power
  - Ventilation Boundary
  - Procedures (Maintenance)

- MDC 2039509801, Local Suppression Indicating Panels, Rev. 1 (Mitigating Systems)
  - Testing Review
  - Energy Needs (Power)
  - Operations

- MDC 2051744901, Unit 2 Diesel Generator Exhaust Stack Replacement (Mitigating Systems)
  - Materials/Replacement Components
  - Equipment Protection
  - Flowpaths
  - Pressure Boundary
  - Structural
  - Process Medium

- MDC 2039000501, Replacement of U2 Emergency Generator Protective Relays (Mitigating Systems)
  - Materials/Replacement Components
  - Control Signals
  - Equipment Protection

- MDC 1049504801, 1AB04X, 1AB05X, and 1AB15X Replacement (Mitigating Systems)
  - Energy Needs
  - Materials/Replacement Components
  - Licensing Basis
  - Testing Review

- MDC 1051602801, P-14 Setpoint Change (Steam Generator Water Level High-High) (Mitigating Systems)
  - Control Signals
  - Licensing Basis
  - Testing Review

- DCP 2049001101, Replace Reactor Vessel Head Conoseals (male flange part) with Westinghouse Core Exit Thermocouple Nozzle Assembly (CETNA), Rev 1.2 (Barrier Integrity)
  - Materials/Replacement Components
  - Pressure Boundary
  - Structural
  - Licensing Basis
  - Testing Review

- DCP 2060523401, Removal of RHR Bypass Lines 2-1201-238-3/4" and 2-1201-239-3/4" from RHR Hot Leg Suction Piping (Barrier Integrity)
  - Materials/Replacement Components
  - Operations
  - Pressure Boundary
  - Structural
  - Process Medium
  - Licensing Basis
  - Failure Modes

The inspectors also reviewed selected CRs and three self-assessments associated with modifications to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors either observed post-maintenance testing or reviewed the test results for the following five maintenance activities to verify that the testing met the requirements of procedure 29401-C, Work Order Functional Tests, for ensuring equipment operability and functional capability was restored. The inspectors also reviewed the test procedures to verify the acceptance criteria was sufficient to meet the TS operability requirements.

- MWO 10526178, Unit 1 train B DG K-146 relay replacement
- MWO 20620854, Unit 2 train B NSCW fan motor #4 replacement
- MWO 10611267, Unit 1 train A DG fuel line replacement
- MWO 20601152, Unit 2 train A RHR pump planned maintenance outage
- MWO 20702782, Investigate failure of Unit 2 train B fan 1 to start

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

The inspectors performed the inspection activities described below for the Unit 2 refueling outage that began on March 4.

- Reviewed the outage risk plan to verify that activities, systems, and/or components which could cause unexpected reactivity changes were identified in the outage risk plan and were controlled.
- Reviewed system alignments to verify that the flow paths, configurations, and alternative means for inventory addition were consistent with the outage risk plan.
- Reviewed reactor coolant system (RCS) pressure, level, and temperature instruments to verify that the instruments provided accurate indication and that allowances were made for instrumentation errors.
- Observed decay heat removal parameters to verify that the system was properly functioning and providing cooling to the core.
- Reviewed the status and configuration of electrical systems to verify that those systems met TS requirements and the licensee's outage risk control plan.
- Reviewed the licensee's plans for changing plant configurations to verify that technical specifications, license conditions, and other requirements, commitments, and administrative procedure prerequisites were met prior to changing plant configurations.

- Reviewed selected control room operations to verify that the licensee was controlling reactivity in accordance with the technical specifications.

The inspectors confirmed that, when the licensee removed equipment from service, the licensee maintained defense-in-depth commensurate with the outage risk control plan for key safety functions and applicable technical specifications, and that configuration changes due to emergent work and unexpected conditions were controlled in accordance with the outage risk control plan.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the following six surveillance test procedures and either observed the testing or reviewed test results to verify that testing was conducted in accordance with the procedures and that the acceptance criteria adequately demonstrated that the equipment was operable. Additionally, the inspectors reviewed the CR database to verify that the licensee had adequately identified and implemented appropriate corrective actions for surveillance test problems.

Surveillance Tests
- 24613-2, Safety Features Sequencer Train A Channel Operational Test and Channel Calibration
- 14666-2, Train A Diesel Generator and ESFAS Test

Containment Isolation Valve Test
- 14342-2, Containment Penetration No. 42 Accumulator Nitrogen Supply Local Leak Rate Test

In-Service Tests (IST)
- 14804-2, Safety Injection Pump Inservice and Response Time Tests
- 14803-1, CCW Pumps and Check Valve IST and Response Time Tests

RCS Leak Detection System
- 14905-2, RCS Leakage Calculation (Inventory Balance)

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors reviewed the facility activation exercise guide and observed the following emergency response activity to verify the licensee was properly classifying emergency events, making the required notifications, and making appropriate protective action recommendations in accordance with procedures 91001-C, Emergency Classifications, and 91305-C, Protective Action Guidelines.

- On February 21, the licensee conducted an emergency preparedness drill involving a loss of reactor coolant and containment breach. The technical support center was activated and the site participated in the exercise.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Performance Indicator Review

a. Inspection Scope

The inspectors sampled licensee submittals for the listed PIs during the period from January 1, 2006, through December 31, 2006, for Unit 1 and Unit 2. The inspectors verified the licensee's basis in reporting each data element using the PI definitions and guidance contained in procedures 00163-C, NRC Performance Indicator and Monthly Operating Report Preparation and Submittal, and Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Indicator Guideline, Revision 4.

Initiating Events Cornerstone
- Unplanned Scrams per 7,000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7,000 Critical Hours

The inspectors reviewed Licensee Event Reports (LERs), Unit 1 and Unit 2 operator log entries, the monthly operating reports, monthly PI summary reports, the maintenance rule database, and NRC inspection reports to verify that the licensee had accurately submitted the PI data.

b. Findings

No findings of significance were identified.

.2 Unit 2 Cooling Water Systems Mitigating System Performance Index (MSPI)

a. Inspection Scope

The inspectors reviewed MSPI data for the Unit 2 Cooling Water systems to verify the licensee had included all unavailability and unreliability hours for reporting of the MSPI.

b. Findings

The inspectors identified that there was an additional 20.5 hours of unavailability time for the Unit 2 NSCW system which had not been included in the Unit 2 Cooling Water Systems MSPI. The licensee determined that the additional hours would not change the color of any previously reported data; however, the licensee did not review the effect the additional hours would have on future MSPI data reporting. On January 17, the licensee reported the MSPI as Green. On February 1, the licensee added the 20.5 hours into their performance indicator data entry program and determined that the Unit 2 Cooling Water Systems MSPI was actually White. On February 7, the licensee submitted a correction to the NRC and reported the MSPI as White.

The NRC has determined that a Severity Level IV violation of 10 CFR 50.9, Completeness and Accuracy of Information, occurred. However, the submission of the incomplete and inaccurate MSPI information was not willful and the incomplete and inaccurate MSPI information was identified within a period of 1 year after the beginning of MSPI data collection, or by April 1, 2007. Therefore, in recognition of this, the ongoing PI development activities, the time constraints to gather and submit historical data, the large volume of data (12 quarters of data) needed to calculate and verify the MSPIs, and the time needed for licensees to familiarize and adjust to the new reporting guidance; the NRC is exercising enforcement discretion pursuant to Section VII.B.6, Violations Involving Special Circumstances, of the Enforcement Policy to refrain from issuing a Notice of Violation in this case (EA-07-095).

4OA2 Identification and Resolution of Problems

Daily Condition Report Review. As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by either attending daily screening meetings that briefly discussed major CRs, or accessing the licensee's computerized corrective action database and reviewing each CR that was initiated.

4OA3 Event Followup

.1 (Closed) LER 05000424/2005003: Feedwater Valve Failure Leads to Reactor Trip. The inspectors reviewed the LER, CR 2005103061, and Event Report 1-2005-02 to verify that the cause of the April 29, 2005, Unit 1 reactor trip was identified and that corrective actions were reasonable. The automatic reactor trip occurred on low-low steam generator water level after the loop 1 MFRV closed unexpectedly during replacement of a failed card in the MFRV control system. The inspectors observed plant parameters for mitigating systems and fission product barriers, evaluated performance of systems and operators, and confirmed proper classification and reporting of the event. No findings of significance were identified.

.2 (Closed) LER 05000424/2005004: Manual Reactor Trip Following Main Feedwater Regulating Valve Failure

a. Inspection Scope

The inspectors reviewed the LER, CR 2005109484, and Event Report 1-2005-04 to verify that the cause of the October 17, 2005, Unit 1 reactor trip was identified and that corrective actions were reasonable. Operators manually tripped the reactor due to rapidly lowering water level in the loop 2 steam generator when the MFRV closed unexpectedly and could not be re-opened from the control room. The MFRV closure was due to an electronic circuit board failure which resulted from moisture intrusion into the MFRV's electro-pneumatic transducer. The transducer was mounted in an inverted manner that did not prevent water ingress and allow drainage. Washdown activities in the vicinity of the valve allowed moisture into the transducer causing the subsequent failure.

b. Findings

Introduction. A Green self-revealing finding was identified for poor work practices associated with the installation of the loop 2 MFRV electro-pneumatic transducer. The transducer was mounted in a manner that did not prevent water ingress and allow drainage. During washdown activities in the vicinity of the MFRV water entered the transducer causing the loop 2 MFRV to close. When the MFRV could not re-opened from the control room, operators manually tripped the reactor.

Description. On September 30, 2003, a Masoneilian model 7000 electro-pneumatic transducer was installed on the loop 2 MFRV. This installation was part of a planned series of transducer replacements for an obsolete model. The Masoneilian instruction manual for the transducer stated that the transducer will operate in any position, but should mounted upright if water ingress effects are to be minimized. The area where this transducer was installed was not protected from water ingress. However, the installation procedure for this transducer did not specify the proper orientation in which the transducer should be mounted. Consequently, the transducer was mounted in a manner that did not prevent water ingress and allow drainage. On October 17, 2005, washdown activities were conducted in the vicinity of the loop 2 MFRV. The licensee determined that water from these washdown activities accumulated in the transducer. The transducer subsequently failed causing the loop 2 MFRV to close. When operators were unable to re-open the MFRV from the control room, the reactor was manually tripped.

Analysis. This finding is more than minor because it affected the procedure quality attribute of the Initiating Events cornerstone in that inadequate installation instructions caused transducer failure which resulted in a manual reactor trip. The finding was determined to be of very low safety significance (Green) because it did not increase the likelihood that mitigation equipment or functions would not be available. This finding is directly related to the complete and accurate procedures aspect of the Human Performance cross-cutting area because the procedure did not incorporate the vendor's recommendations.

Enforcement. The inspectors determined that the finding did not represent a violation of regulatory requirements because it only involved non-safety related plant equipment. This finding will be tracked as FIN 05000424/2007002-01, Inadequate Work Instructions for Maintenance Resulted in the Failure of Unit 1 Loop 2 Main Feedwater Regulating Valve.

.3 (Closed) LER 05000425/2005-001: High Flux at Shutdown Alarm Inoperable – A Condition Prohibited by the Technical Specifications. On June 2, 2005, while performing a surveillance licensee personnel identified that both trains of the Unit 2 solid state protection system (SSPS) were in "input error inhibit" which rendered the High Flux at Shutdown Alarm inoperable. Subsequent investigation determined that this condition had existed since May 25, 2005, which exceeded the allowed out-of-service time specified in Technical Specification 3.3.8, Condition B. The cause of the event was determined to be the improper use of an operating procedure to place the solid state protection system (SSPS) circuit in TEST. There was no specific guidance for performing this activity; however, the operators failed to consult with either Instrumentation and Control or Engineering personnel to ensure the appropriate actions were bring taken when they were unable to find clear procedural guidance. The operators then used portions of selected sections of the system operating procedures to place the system in TEST. The procedure sections used were designed to de-energize the SSPS circuitry rather than placing both trains in TEST which resulted in the alarm circuitry being rendered inoperable for approximately 8 days. This condition was identified during surveillance testing of the neutron flux instrumentation. Once the condition was identified, the High Flux at Shutdown Alarm circuit was promptly returned to service. The enforcement aspects of this violation are discussed in section 4OA7.

.4 Unit 2 Unidentified Leakage and Notification of Unusual Event.

On March 9, Unit 2 was in a refueling outage. The plant was depressurized at a temperature of approximately 90 degrees F and reactor coolant system (RCS) inventory was maintained by balancing charging and letdown. The reactor level indication system (RVLIS) was out of service for maintenance. The operators were reducing RCS inventory in preparation for removal of the reactor vessel head. The operators then intended to reduce level to 197 ft and stabilize while one core exit thermocouple nozzle assembly was removed to provide additional vent capacity. Once the RCS level was reduced to the top of the reactor vessel head (200 ft) the head vent was opened. Operators noted that in order to control RCS level, charging was approximately 50 gpm in excess of letdown. Additionally, gas was noted to be exhausting from the reactor vessel head vent. There were no auxiliary building sump alarms, abnormal increases in containment sump levels, and no surge tank high level alarms for any systems connected to the RCS noted by the operators. The licensee entered Abnormal Operating Procedure 18004-C, Reactor Coolant System Leakage. Based on the mismatch between charging and letdown, the shift manager determined that he had unidentified RCS leakage in excess of 10 gpm, and in accordance with the site emergency plan, declared a Notification of Unusual Event (NOUE). The gas was allowed to continue venting from the reactor vessel head, charging was maintained as necessary to control RCS level at 197 feet, and the mismatch gradually reduced until charging and letdown matched.

The licensee preliminarily attributed the mismatch between charging and letdown to a gas pocket which formed in the vessel head since the plant was depressurized. This gas pocket was vented when the reactor vessel head vent was opened. The licensee performed an engineering evaluation to confirm that assessment and terminated the NOUE.

During this event inspectors determined that safety equipment operated as designed, and plant personnel adequately controlled plant parameters. The decision to declare a NOUE was reviewed by regional NRC emergency preparedness inspectors and documented in Inspection Report 05000424,425/2007501.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/150, Rev. 3, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009) (Unit 2)
(Closed) Unresolved Item 05000424/2006005-03, Basis for Reactor Pressure Vessel Head Temperature used in Susceptibility Category Calculation

a. Inspection Scope

The inspectors reviewed the licensee's activities associated with the NDE of the reactor pressure vessel head (RPVH) penetration nozzles and the visual examination to identify potential boric acid leaks from pressure-retaining components above the RPVH in response to NRC Bulletins 2001-01, 2002-01, 2002-02, and the first revision of NRC Order EA-03-009 Modifying Licenses dated February 20, 2004 (hereafter referred to as the NRC Order).

The inspectors' review of the NDE of RPVH penetration nozzles included independent observation and evaluation of UT and ET examinations (for both data acquisition and analysis), review of NDE procedures, and personnel qualifications and training. The inspectors also held interviews with contractor representatives (Wesdyne) and other licensee personnel involved with the RPVH examination. The activities were reviewed to verify licensee compliance with the NRC Order and to gather information to help the NRC staff identify possible further regulatory positions and generic communications.

The inspectors reviewed a sample of the results from the volumetric UT and surface ET examinations of RPVH penetration nozzles. Specifically, the inspectors reviewed or observed the following:

- Observed in-process UT/ET data acquisition scanning of RPVH penetration nozzles 24 and 64 (one nozzle with thermal sleeve, one with open housing)
- Reviewed the UT/ET data with the Level III analyst for RPVH nozzles 10, 17, 22, 24, 25, 41, 48, 61, 74, 77, and 29 (this included nozzles both with and without thermal sleeves).

- Reviewed the results of the UT examination performed to assess for leakage into the annulus (interference fit zone) between the RPVH penetration nozzle and the RPVH low-alloy steel for all penetration numbers listed in the previous bullet
- Reviewed the procedure and results for the visual exam performed to identify potential boric acid leaks from pressure-retaining components above the RPVH
- Reviewed the RPVH susceptibility ranking and calculation of effective degradation years (EDY), including the basis for the RPVH temperature used in the calculation

b. Observations and Findings

In accordance with the requirements of TI 2515/150, the inspectors evaluated and answered the following questions:

1) Were the examinations performed by qualified and knowledgeable personnel?

Yes. All personnel involved with the RPVH inspections were appropriately qualified in accordance with the ASME Code, and most far exceeded the minimum requirements for experience and training hours. The contractor (Wesdyne) personnel responsible for equipment manipulation, data acquisition, and data analysis were comprised of a multi-national group of people who frequently perform these types of inspections worldwide.

2) Were the examinations performed in accordance with demonstrated procedures?

Yes. The Vogtle Unit 2 RPVH has 61 control rod drive mechanism (CRDM) nozzles with thermal sleeves, 17 penetrations with open housings (including 4 instrument column nozzles), and 1 vent nozzle, for a total of 79 nozzles. All nozzles, excluding the vent, were subject to both UT and ET remote automated examination. There were two different types of probes primarily used to perform the exams. One type (the Gap Scanner probe) was used for sleeved penetrations, and the other type (the 7010 probe) was used for the open housing penetrations. Both types of probes have UT transducers and ET coils so that the exams can be performed in coincidence, however the UT and ET exams were addressed under separate procedures. Procedures WDI-UT-010 and WDI-UT-013 were used for UT data acquisition and analysis, respectively. Similarly, procedures WDI-ET-003 and WDI-ET-004 were used for ET. These procedures were implemented for both types of probes, except that there was a separate procedure used for the ET acquisition using the Gap Scanner (WDI-ET-008).

The inspectors verified that the above procedures were implemented during the RPVH examinations. Further, the inspectors verified that these procedures were used during Wesdyne's demonstration for EPRI's Materials Reliability Program (MRP) to show flaw detection capability in RPVH penetrations. The results of this demonstration, as reported in MRP-89, were reviewed. It should be noted that the procedures have undergone revisions since initial demonstration to the MRP, however technical justifications for any change to an essential variable have been documented.

A third type of probe, the Grooveman, was used on penetrations 7 and 9 to ensure that full inspection coverage was obtained per the NRC Order. This probe performs an ET exam of the surface of the J-groove weld. Procedure WDI-ET-002, which is specific to the Grooveman probe, was implemented for these two exams.

The RPVH vent line penetration received only ET examination via manual acquisition. The exam was performed in two steps, one for the tube inner diameter and one for the surface of the J-groove weld. These activities were performed in accordance with Wesdyne procedures WDI-STD-114 and WDI-STD-101.

3) Was the examination able to identify, disposition, and resolve deficiencies?

Yes. All indications of cracks or interference fit zone leakage are required to be reported for further examination and disposition. Based on observation of the examination process, the inspectors considered that deficiencies would be appropriately identified, dispositioned, and resolved. UT indications associated with the geometry of the examined volume were identified in several penetration tubes. ET indications associated with minor surface scratches due to centering pad wear were also identified in several tubes. No indications exhibited crack-like characteristics and were appropriately dispositioned in accordance with procedures.

4) Was the examination capable of identifying the primary water stress corrosion cracking (PWSCC) and/or RPVH corrosion phenomena described in the NRC Order?

Yes. The NDE techniques employed for the examination of RPVH nozzles had been previously demonstrated under the EPRI MRP/Inspection Demonstration Program as capable of detecting PWSCC type manufactured cracks as well as cracks from actual samples from another site. Based on the demonstration, observation of in-process examinations, and review of NDE data, the inspectors determined that the licensee was capable of identifying PWSCC and/or corrosion as required by the NRC Order.

5) What was the physical condition of the RPVH (e.g. debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

A 100% bare metal visual (BMV) examination was not performed this refueling outage, therefore the condition of the RPVH under the insulation could not be assessed.

6) Could small boron deposits, as described in NRC Bulletin 2001-01, be identified and characterized?

The periodic BMV examination required by the NRC Order was not performed this refueling outage.

7) What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

There were no identified examples of RPVH penetration cracks, leakage, material deficiencies, or other flaws that required repair. As discussed previously, there were some UT indications at J-groove welds that were dispositioned as metallurgical/geometric indications (not service related). These indications were likely due to weld repairs performed during initial RPVH fabrication. It should also be noted that this volumetric examination served as a baseline inspection since this was the first volumetric exam of the Vogtle Unit 2 RPVH.

8) What, if any, impediments to effective examinations, for each of the applied methods, were identified (e.g., centering rings, insulation, thermal sleeves, instrumentation, nozzle distortion)?

The penetration nozzles with thermal sleeves and centering pads did not impede access to the examination area. However, there was difficulty in obtaining full UT data coverage for penetrations 7 and 9 using the GapScanner probe. The ultrasonic transducers could not maintain contact with the tube surface on these penetrations in the areas where there were surface scratches due to centering pad wear. However, the ET probe on the GapScanner was able to obtain quality data in these areas. Therefore, the Grooveman probe was used to obtain surface data on the J-groove weld. This combination of surface exams meets the requirements of paragraph IV.C.(5)(b)(ii) of the NRC Order, which is an acceptable alternative to full UT volumetric examination.

Concerning examination coverage, the NRC Order requires that each tube's volume is inspected from a minimum of 2 inches above the highest point of the J-groove weld to 2 inches below the lowest point of the J-groove weld (1 inch with proper stress analysis). Due to the physical dimensions of some penetration tubes, the lower coverage could not always be achieved. Therefore, the licensee requested and subsequently received an NRC authorized Relaxation of Requirements based on a crack growth evaluation. Letter NL-06-1986, dated August 30, 2006, documents this granted Relaxation and the NRC staff's safety evaluation (ADAMS Accession Number ML062360585). The inspectors determined that the minimum examination coverages required by the Relaxation were met.

9) What was the basis for the temperature used in the susceptibility ranking calculation?

NRC Order EA-03-009 requires that licensees calculate the effective degradation years (EDY) of the reactor pressure vessel head (RPVH) to determine its susceptibility category, which subsequently determines the scope and frequency of required RPVH examinations. The operating temperature of the RPVH is an input to this calculation. Therefore, an incorrect temperature input could result in placing the RPVH in an incorrect susceptibility category. The licensee uses the cold leg temperature (with an added 3°F conservatism) as the RPVH temperature. This input value is 560°F. An unresolved item was opened during the Unit 1 RPVH inspection in October 2006 due to the unavailability of documentation to validate the basis for using this temperature in the calculation.

During this inspection, the licensee provided documentation from Westinghouse describing the methodology and results of a calculation which determined the mean bulk fluid temperature in the RPVH region for both Vogtle Units 1 and 2. Westinghouse developed an analytical model for the Westinghouse fleet of plants based on flow rates from both the head cooling spray nozzles and the bypass flow through the guide tubes and support columns. This model was verified experimentally during 1/5 scale model testing. It was further validated when actual temperature measurements were obtained in vessel heads similar to the Vogtle units which showed that the average temperature was indeed approximately equal to the cold leg temperature. Specifically for both Vogtle units, the actual cold leg temperature is 556.9°F +/- 0.5°F. The Westinghouse model calculated a RPVH temperature of 558.9°F. However, Vogtle uses a conservative value of 560°F in their effective degradation years calculation. Based on this information, there is an adequate basis for use of 560°F as the RPVH operating temperature in the susceptibility category calculation as required by the NRC Order. Therefore, URI 05000424/2006005-03, Basis for Reactor Pressure Vessel Head Temperature used in Susceptibility Category Calculation, is closed.

10) During non-visual examinations, was the disposition of indications consistent with the NRC flaw evaluation guidance?

    There were no indications considered to be flaws found during the RPVH examination.

11) Did procedures exist to identify potential boric acid leaks from pressure-retaining components above the RPVH?

    Yes. Procedure 84008-C, RCS Alloy 600 Material Inspection Program, exists to meet several requirements of the NRC Order, including the inspection to identify potential boric acid leaks from pressure-retaining components above the RPVH. The inspectors determined that the procedure implementation met the requirements of the NRC Order. The inspectors also reviewed the inspection results for this outage and found that no indications of boric acid leakage from pressure-retaining components above the RPVH were identified.

12) Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components above the RPVH?

    There were no indications of leakage found during this outage.

4OA6 Meetings, Including Exit

.1 Exit Meeting

On April 12, the resident inspectors presented the inspection results to Mr. T. Tynan and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

On April 11, the Chief of Reactor Projects Branch 2 and the Senior Resident Inspector assigned to the Vogtle Electric Generating Plant (VEGP) met with Southern Nuclear Operating Company to discuss the NRC's Reactor Oversight Process (ROP) and the NRC's annual assessment of VEGP safety performance for the period of January 1, 2006 - December 31, 2006. The major topics addressed were: the NRC's assessment program and the results of the VEGP assessment. A listing of meeting attendees and information presented during the meeting are available from the NRC's document system (ADAMS) as accession number ML071020182. ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html*.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Technical Specification 3.3.8, Condition B requires that two channels of the High Flux at Shutdown Alarm be operable when the reactor plant is in mode 5. Contrary to this requirement, on June 2, 2005, when the plant was in mode 5, licensee personnel performing a surveillance identified that both trains of the Unit 2 SSPS were in "input error inhibit" which rendered the High Flux Alarm at Shutdown circuit inoperable. This finding is of very low safety significance because this finding does not increase the likelihood of a loss of RCS inventory, does not degrade the licensee's ability to terminate a leak or add RCS inventory, or degrade the licensee's ability to recover decay heat removal once it is lost. The licensee has documented this issue in condition report 2005103989.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
R. Brown, Training and Emergency Preparedness Manager
C. Buck, Chemistry Manager
W. Copeland, Licensing Supervisor
R. Dedrickson, Plant Manager
K. Dyar, Security Manager
I. Kochery, Health Physics Manager
D. Lambert, Design Mods Supervisor
J. Robinson, Operations Manager
S. Swanson, Engineering Support Manager
T. Tidwell, Plant Mods Supervisor
T. Tynan, Site Vice-President
J. Williams, Site Support Manager

NRC personnel:
S. Shaeffer, Chief, Region II Reactor Project Branch 2

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000424/2007002-01	FIN	Inadequate Work Instructions for Maintenance Resulted in the Failure of Unit 1 Loop 2 Main Feedwater Regulating Valve (Section 4OA3.2)

Closed

05000424/2005003	LER	Feedwater Valve Failure Leads to Reactor Trip (Section 4OA3.1)
05000424/2005004	LER	Manual Reactor Trip Following Main Feedwater Regulating Valve Failure (Section 4OA3.2)
05000425/2005-001	LER	High Flux at Shutdown Alarm Inoperable – A Condition Prohibited by the Technical Specifications. (Section 4OA3.3)
2515/150 (Unit 2)	TI	Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009). (Section 4OA5)
05000424/2006005-03	URI	Basis for Reactor Pressure Vessel Head Temperature used in Susceptibility Category Calculation. (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R02: Evaluation of Changes, Tests, or Experiments

Full Evaluations
1019003501, AFW Controls Upgrade, Rev. 0
2004058 AFT-0014, Flow Transmitter, Rev. 1
DCP C009006701, Fuel Building Cask Crane Enhancements
LDCR 2006009, Unit 2 North Hatch Hoist and Trolley (ACCW HX 1 room), Rev. 0
LDCR 2003034, Revise FSAR to show latest approved AFW capabilities in table 10.4.9-3, Rev. 0.
EVAL-04-124, Primary System Zinc Addition for Vogtle Unit 2, Rev 0
EVAL-06-20, Vogtle Units 1 and 2 RHR Bypass Line and Valve Bonnet Depressurization Line Removal, Rev 0

Screened Out Items
MDC 01-V1M011 (101951101) RHR Discharge Valve Hardstop, Rev. 1
MDC 1039509701, Removal of Jumper on LSIPS to Prevent Inadvertent Actuation, Rev. 1
MDC 04-V1M045 (1049504501), Unit 1 Essential Chilled Water System “A” Train Flow Orifice Replacement, Rev. 1 (one FCR)
MDC 2040377201, Unit 2 ACCW System Pressure Reduction, Rev. 1
MDC C039500901, Isolated Phase Bus Secondary Side Wiring
MDC 1040424501, Diesel Generator Exhaust Stack Replacement
MDC 2051744901, Unit 2 Diesel Generator Exhaust Stack Replacement
MDC 2039000501, Replacement of U2 Emergency Generator Protective Relays
MDC 1039509701, Remove Jumpers on LSIPs to Prevent Inadvertent Actuation, Rev. 1
MDC 1049504801, 1AB04X, 1AB05X, and 1AB15X Replacement, Rev. 1
LDCR 2005037, Auxiliary Feedwater Pump Discharge Valve Operation
MDC 2060971501, 2T0412 TAVG Auctioneer and Deviation Alarm
MDC C029001701, Remove and Replace Inverter 2BD1L12
MDC A060207801, Replacement of 1K1, 1K2 relay in TSC Chiller
MDC 1051602801, P-14 Setpoint Change (Steam Generator Water Level High-High)
DCP 2049001101, Replace Reactor Vessel Head Conoseals (male flange part) with Westinghouse Core Exit Thermocouple Nozzle Assembly (CETNA), Rev 2

CRs: 2005111343, 2006104331, 2006114010, 2002003346, 2005100574

Procedures
DC-1020, Rev 4, Spent Fuel Cask Bridge Crane
DC-2109, Ver 7, Fuel Handling Building
DC-2301, Fire Protection Water System, Rev. 9
DC-1022, Miscellaneous Cranes and Hoists, Rev. 3

Work Orders
10302560, 1HV0607
10302550, 1HV0606

Calculations
Calculation C4C1217V07, Typical ACCW Operating Pressures, 7/19/01
X4C1302S12, AFW Pump Discharge Line Orifice Sizing, Rev. 1

Other Documents
ABN 1X6AA06-00114, 02TQ097 Fisher Letter, 1/10/03
RER 2003-V0336, ACCW piping System, 4/19/04
REA, 00-VAA622, AFT-0014 Downgrade, 5/16/2000
REA 98-VAA616, Orems Flow Transmitters, 10/24/98
LCDR 2004058, FSAR Table 11.5.2-5, Rev. 1
Supply Requisition, GP25-002496, 7/7/00
Purchase Order 7046312, AF-0014, 7/28/00
Vogtle Electric Generating Plant Unit 1 & 2 Combined Master List of Safety-Related Equipment Located in a Harsh environment, Rev. 21 [Unit 1]
DC-1822, Rev 5, Isophase Bus System Design Criteria
RER 2002-0316, Evaluation of PT-3C Connection Alternatives
UFSAR, Rev 14, Sections 1.2.10.6 and 9.1.5.2
REA 00-VAA090, Hoist Conceptual Design
LDCR 2003061, Spent Fuel Cask Bridge Crane Enhancements
LDCR 2005012, Zinc Addition to the Unit Two Reactor Coolant System, Rev 1
X2CK04.55.5, Addendum 1, Aux Bldg Cut Rebar Evaluation (Levels 1,2 & 3)
AX4AL03-81-2, Harnischfeger P&H Bulletin ED-5-3, Disc Brake Rectifier
93230-C, Rev 11, Spent Fuel Cask Bridge Crane Operating Instructions
RER 2002-0201, Remove Jumpers on LSIPs to Prevent Inadvertent Actuation
Specification X4AL05, Miscellaneous Hoists, Rev.6
RER 2003-0081, Revise FSAR to show latest approved AFW capabilities, Rev. 0

Section 1R04: Equipment Alignment
Procedures
11610-1, Auxiliary Feedwater Alignment
13145-1, Diesel Generators
11145-1, Diesel Generator Alignment
11701-1, Boric Acid System Alignment
13701-1, Boric Acid System
13006-1, Chemical and Volume Control System
11006-1, Chemical and Volume Control System Alignment

Drawings
1X4DB161-1, P&I Diagram, Auxiliary Feedwater
1X4DB161-2, P&I Diagram, Auxiliary Feedwater
1X4DB161-3, P&I Diagram, Auxiliary Feedwater
1X4DB170-2, Diesel Generator System Train B
1X4DB118, Chemical and Volume Control System
1X4DB116-1, Chemical and Volume Control System
1X4DB111, Reactor Coolant System
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92855-2, Zone 155 Auxiliary Feedwater Pumphouse Train B Fire Fighting Preplan
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92857A-2, Zone 157A Auxiliary Feedwater Pumphouse Train C Fire Fighting Preplan
92857B-2, Zone 157B Auxiliary Feedwater Pumphouse Train C Fire Fighting Preplan
92860A-1, Zone 160A NSCW Pumphouse Train A Fire Fighting Preplan
92860B-1, Zone 160B NSCW Pumphouse Train B Fire Fighting Preplan
92791-2, Zone 91 Control Building Level A Fire Fighting Preplan
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92757A-1, Zone 57A Control Building Levels B, A, 1, 2, and 3 Fire Fighting Preplan
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92879-1, Zone 179 Control Building Level 3 Fire Fighting Preplan
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92825A-1, Zone 125A Control Building Level 3 Fire Fighting Preplan
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Procedures
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Section 1R22: Surveillance Testing

Procedures

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